May 10, 2004

MEMORANDUM TO: Ledyard B. Marsh, Director
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

FROM: Charles E. Ader, Director */RA/*
Division of Risk Analysis and Applications
Office of Nuclear Regulatory Research

SUBJECT: TRANSMITTAL OF FINAL ASP ANALYSIS

This memorandum provides seven Accident Sequence Precursor (ASP) analyses of operational events which occurred at various plants during fiscal year 2001. These events are being issued as final analyses.

As stated in SECY-03-0049 (dated March 31, 2003), there have been delays in issuing these analyses due to the ASP program's focus on several more complex and potentially risk important events. We have developed, and are currently implementing, a plan for the completion of the backlog of ASP analyses. Consistent with our plan, an ASP analysis will be issued as final for non-controversial, lower risk precursor in which the ASP results are consistent with the results from the Significance Determination Process's (SDP's) final evaluation of the same condition. Elimination of the review and comment resolution cycle for these events will reduce burden for NRC staff and the licensee.

Transmittal to licensees requested. We are requesting NRR/DLPM to send the final ASP analyses to the appropriate licensees for information. The analyses and a transmittal letter will be provided separately to the NRR ASP Program liaison (Stacey Rosenberg).

Final ASP analyses to be transmitted. Attachment 1 summarizes the final analyses of the following events and conditions:

- Turbine-driven auxiliary feedwater pump unavailable for one year at Calvert Cliffs Unit 1 in May 2001(LER 317/01-001). ASP analysis calculated the mean change in core damage probability (ΔCDP) of 7.3×10^{-6}.

- High pressure coolant injection system water hammer event at Dresden Unit 3 in July 2001(LER 249/02-005). ASP analysis calculated a mean ΔCDP of 3.1×10^{-6}.

- Smoke detectors not properly installed at Palisades in September 2001 (Inspection Report 20-255/01-08). ASP analysis calculated a mean ΔCDP of 1.4×10^{-6}.

- Emergency diesel generator unavailable for long mission times for several months and inoperable for 10 days for repairs at Fermi Unit 2 in March 2001 (LER 341/01-001). ASP analysis calculated a mean ΔCDP of 3.0×10^{-6}.

- Manual scram following MSRV failing open due to erosion and oxidation of the first stage pilot valve disc seating area at Limerick Unit 2 in February 2001 (LER 353/01-001). ASP analysis calculated a mean ΔCDP of 3.2×10^{-6}.

- Inadequate fire protection and procedures for the north switchgear room at Arkansas Nuclear One, Unit 1 in August 2001 (Inspection Report 50-313/01-06). ASP analysis calculated a mean ΔCDP of 4.0×10^{-6}.

- Reactor scram due to undervoltage protective circuit actuation on Division 1 ESF bus at La Salle Unit 2 in September 2001 (LER 374/01-003). ASP analysis calculated a mean CCDP of 1.0×10^{-5}.

Sensitive information. The detailed ASP analyses are classified as "SENSITIVE - NOT FOR PUBLIC DISCLOSURE." This classification is based on the guidance provided by the EDO in the memorandum to the Commission (dated April 4, 2002), concerning the release of information to the public that could provide significant assistance to support an act of terrorism. In particular, Criteria 1 was determined to be applied to ASP analysis reports:

> Plant-specific information, generated by NRC, our licensees, or our contractors, that would clearly aid in planning an assault on a facility. An example might be drawings depicting the location of certain safety equipment within plant buildings. Examples may include portions of Final Safety Analysis Reports (FSARs), Individual Plant Examination (IPE) material, and other risk and facility vulnerability information.

This classification could change in the future based on revised Agency guidance and office (NRR and RES) procedures in response to the Staff Requirements Memorandum, "Staff Requirements - COMSECY-02-0015 - Withholding Sensitive Homeland Security Information From the Public," dated April 4, 2002. Future changes in the transmittal of ASP analyses will be coordinated with the NRR ASP Program liaison. Attachment 1 is provided to summarize the ASP analyses without disclosing any plant-specific risk and facility vulnerability information. The detailed ASP analyses, referenced in the Attachment, are sensitive and cannot be released to the public.

If you have any questions about the individual analyses, please contact the reviewer for that analysis. For questions concerning the transmittal letter or the ASP Program, please call Gary DeMoss (415-6225).

Attachment:
Summaries of Final ASP Analyses

MEMORANDUM DATED: 5/10/04

SUBJECT: TRANSMITTAL OF FINAL ASP ANALYSIS

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Attachment 1

SUMMARIES OF FINAL ASP ANALYSES

Unavailability of the turbine-driven auxiliary feedwater pump for one year, Calvert Cliffs 1 (May 2001)

This is the final ASP Program analysis of an operational condition which was discovered at Calvert Cliffs Unit 1 on May 16, 2001 The condition was documented in LER 317/01-001, dated July 13, 2001 and Inspection Report 50-317/01-09, dated August 24, 2001 and in the final Significance Determination for this inspection report, Enforcement Action 01-206.

Condition summary. This analysis involves a failure of the auxiliary feedwater turbine-driven pump (No. 11 AFWTDP) turbine outboard bearing during a surveillance test (failure to run). At the time of discovery, the plant was in Mode 1 with reactor at 100 percent power. The No. 11 AFWTDP was declared inoperable. The licensee stated in their LER that the AFWTDP had a total fault exposure time of over one year, even though they had accumulated 10.5 hours of successful run time during that year. The cause of the problem was excessive application of sealant on the bearing oil housing such that the sealant could contaminate the oil.

This condition resulted in a mean increase in the core damage probability (ΔCDP) of 7.3×10^{-6} for the internal events assessment and a ΔCDP of 5.0×10^{-6} as the contribution from external events. The uncertainty about the internal events mean is: 5% bound, 2.1×10^{-6} and 95% bound, 1.8×10^{-5}.

SDP/ASP comparison. The SDP final analysis stated that the internal events conditional assessment was 8.2×10^{-6} which is similar to the ASP ΔCDP of 7.3×10^{-6}. The final SDP ΔCDP for external events was 1.7×10^{-5}, which is somewhat higher than the ASP ΔCDP of 5.0×10^{-6}. Overall, the ASP results are consistent with the SDP finding of YELLOW, with some technical differences in the analysis, which are described below.

The differences between the ASP external events conditional assessment (ΔCDP of 5.0×10^{-6}) and the SDP assessment (where 1.7×10^{-5} appears to be the contribution due to external events) are that the ASP analysis used the SPAR model with seismic and fire event frequencies and the SDP analysis used the licensee value of 2.5×10^{-5} without specific analysis.

The ASP analysis can be found at ML041210891. If you have any questions about the analysis, please contact Jim Houghton (415-6353).

Inoperable high pressure coolant injection system after water hammer event at Dresden (November 2001)

This is the final ASP Program analysis of an operational condition which was discovered at Dresden Nuclear Power Station (DNPS) on July 5, 2001 The condition was documented in LER 249/02-005, dated February 14, 2003 and Inspection Report 50-249/01-21, dated November 16, 2001, and in the final Significance Determination for this inspection report, Enforcement Action 02-264.

Condition summary. This analysis involves an operating condition which occurred over a 87-day period when a potential for a water hammer condition rendered the High Pressure Coolant Injection (HPCI) system inoperable. The condition affected only the HPCI system operability on demand following a transient or a loss of offsite power event. The ASP analysis estimated a mean importance of 3.1×10^{-6}, with 5% uncertainty bound of 8.2×10^{-7} and 95% uncertainty bound of 7.3×10^{-6}.

SDP/ASP comparison. The Phase 3 risk evaluation of the SDP assessment estimated an increase in core damage frequency (CDF) of 3.49×10^{-6} for internal initiating events, and a WHITE finding. No significant differences in SDP and ASP results were found for internal initiating events. Dominant risk contributors in the ASP analysis and the SDP assessment were the same.

The SDP used the licensee's fire PRA to estimate an increase in CDF of 4.2×10^{-6} due to postulated fires. The SDP also used the licensee's seismic PRA results and found that seismic events were not a significant contribution to total increase in CDF. The total increase in CDF due to internal events and fires was 7.2×10^{-6}. Overall, the combined effect from internal events and fire events did not result in exceeding the WHITE threshold. The ASP analysis did not calculate the contributions from external events.

The ASP analysis can be found at ML041210885. If you have any questions about the analysis, please contact Erul Chelliah (415-6186).

<u>Smoke detectors not properly installed at Palisades, (September 2001)</u>

This report is the final ASP Program analysis of an operational condition which was discovered at Palisades Nuclear Power Plant. The condition was documented in Inspection Report 50-255/01-08, dated September 11, 2001 and in final Significance Determination for this inspection report, dated October 26, 2001.

Condition summary. The inspection report 50-255/01-08 described the failure to properly install and locate smoke detectors in the northwest portion of the cable spreading room in accordance with the National Fire Protection Code. At that location, there was a ventilation exhaust that would draw smoke from a fire in equipment important to safety and would delay smoke propagation to the detector. The ASP analysis calculated a mean increase in core damage probability of 1.4×10^{-6}. The uncertainty about the mean is: 5% bound, 1.3×10^{-7} and 95% bound, 4.9×10^{-6}.

SDP/ASP comparison. A comparison between SDP results and the ASP results indicated similar results but with some differences in the analysis. The SDP analysis assumed that the postulated fire would propagate to all trains of safe shutdown equipment in the cable spreading room, requiring use of alternate safe shutdown system and resulted in a WHITE finding. The ASP analysis reviewed the licensee's analysis and agreed that the postulated fire would be limited to one train of safe shutdown equipment, but arrived at similar results.

The ASP analysis can be found at ML041210869. If you have any questions about the analysis, please contact Jim Houghton (415-6353).

Diesel Generator 14 unavailable for long-mission events for 8 months and inoperable for greater than the Technical Specification allowed 7 Days at Fermi (March 2001)

This is the final ASP Program analysis of an operational condition which was discovered at Fermi on July 5, 2001 The condition was documented in LER 341/01-001, dated March 28, 2001, Inspection Report 50-341/01-06, dated April 27, 2001, and Inspection Report 50-341/01-09, dated July 25, 2001.

Condition summary. On March 21, 2001, a 24-hour endurance surveillance test run on emergency diesel generator (EDG) 14 was started. At about 12 hours into the run, the EDG 14 outboard bearing temperature was 228°F and rising. The EDG was manually tripped, but about 1 minute later, there was a fire on the generator outboard bearing housing. The bearing had catastrophically failed due to a lack of lubrication in the bearing housing. The licensee determined that the EDG 14 oil sight glass and level mark were positioned too low. The diesel was subsequently repaired, tested, and returned to service on March 31, 2001. The ASP analysis point estimate for increase in core damage probability is 3.0×10^{-6}.

SDP/ASP comparison. The SDP final analysis stated that the conditional assessment was 3.8×10^{-6} and a WHITE finding, which is similar to the ASP ΔCDP of 3.0×10^{-6}. The SDP and the ASP analysis agreed on the major assumption used for analysis concerning the time that the EDG would not have successfully completed long missions. The time from the last successful test until the EDG was returned to service was about 17 months. Because there is uncertainty about when the failure cause actually rendered the EDG unable to perform its design function, the time for the condition assessment is taken as one-half of the 17-month interval (October 22, 1999, to March 21, 2001), or 8.5 months (6185 hours). The SDP did not provide calculational details, but came up with essentially the same result.

The ASP analysis can be found at ML041210788. If you have any questions about the analysis, please contact Eli Goldfiez (415-5539).

Manual scram following MSRV failing open and MSRV likely to be stuck open at Limerick (February 2001)

This is the final ASP Program analysis of an operational condition which was discovered at Limerick on February 23, 2001. The condition was documented in LER 353/01-001, dated April 24, 2001 and Inspection Report 50-353/01-11, dated December 7, 2001, and in the final Significance Determination for this inspection report, Enforcement Action 01-293.

Condition Summary: On February 23, 2001, operators at Limerick 2 were performing a planned shutdown to repair the 2N and 2M main steam relief valves (MSRVs), which had been leaking for several months. During power reduction with the unit at 92% power, the 2N MSRV inadvertently lifted and remained open for two separate periods. Following the first opening of the 2N MSRV, a manual reactor scram was initiated approximately 2 minutes after the opening per the technical specifications. During the expected postscram level shrink, the (1) reactor protection system and (2) primary containment and reactor vessel isolation control system both received a valid actuation signal. Two loops of the residual heat removal (RHR) system were manually placed into service in the suppression pool cooling mode. The main steam isolation valves were closed to control the reactor depressurization and cooldown rate. However, the

rapid depressurization resulting from the MSRV opening caused a violation of the technical specifications' maximum cooldown rate of 100 °F per hour. The first period of the 2N MSRV remaining open lasted 3 hours and 51 minutes with reactor pressure dropping from 1,040 psig to 122 psig.

Forty minutes after the 2N MSRV closed the first time, the valve again inadvertently opened for a 35-minute period, during which reactor pressure dropped from 206 psig to 122 psig

The ASP analysis performed a condition assessment for the 81 days in which the MSRV was in a condition that it would stick open if it lifted, and calculated point estimate for increase in core damage probability of 5.2×10^{-6}.

SDP/ASP Comparison: The SDP performed a Phase 2 analysis and determined that the event was WHITE. (SDP Phase 2 does not produce a quantitative risk estimate.) The ASP calculation of 5.2×10^{-6} confirms the SDP calculation.

The ASP analysis can be found at ML041210875. If you have any questions about the analysis, please contact Eli Goldfeiz (415-5539).

<u>Inadequate fire protection and procedures for the North Switchgear Room, Fire Zone 99-M at Arkansas Nuclear One, Unit 1, March 2001</u>

This is the final ASP Program analysis of an operational condition which was discovered at Arkansas Nuclear One, Unit One on August 20, 2001 The condition was documented in Inspection Report 50-313/01-06, dated August 20, 2001, and in the final Significance Determination for this inspection report, Enforcement Action 03-016.

Condition Summary: The inspection report indicated that the North Switchgear Room (Fire Zone 99-M) power and control cables associated with redundant trains of equipment, credited in the licensee's hazards analysis for achieving and maintaining hot shutdown conditions, were not protected from fire damage by one of the methods specified in Appendix R, Section III.G.2. In lieu of providing the Appendix R protection from damage, the licensee credited manual actions to remotely operate equipment necessary for achieving and maintaining hot shutdown. The scenario modeled by the ASP is a postulated energetic fire in Fire Zone 99-M A-4 4kV switchgear and is assumed to propagate to the balance of equipment unless successfully suppressed by the fire brigade.

The ASP analysis calculated a mean ΔCDP of 4.0×10^{-6}. The 5% uncertainty bound is 9.6×10^{-8} and 95% uncertainty bound is 1.6×10^{-5}. This uncertainty range is larger than for most ASP events because the accident sequences are overwhelmingly dominated by human failure events.

SDP/ASP Comparison: The risk significance associated with this condition was preliminarily estimated by the SDP to be in the range of 7E-6 to 2E-5, "greater than GREEN." The SDP stated that additional increases in the CDF were warranted due to the existence of other fire zones in the plant which the licensee credited the use of operator action in lieu of meeting 10 CFR Part 50, Appendix R, Section III.G.2 Separation. After additional analysis and comments

by the licensee, the NRC concluded that the condition should be classified as WHITE. Thus, the ASP and SDP findings are similar.

The ASP analysis can be found at ML041210901. If you have any questions about the analysis, please contact Jim Houghton (415-6353).

Reactor scram due to undervoltage protective circuit actuation on Division 1 ESF bus at La Salle Unit 2 (September 2001)

This event was documented in licensee event report (LER) 374/01-002, event date September 8, 2001.

Condition Summary: This reactor was manually tripped due to decreasing reactor pressure vessel water level caused by complications with the feedwater pump controllers when power was lost to the 4 kV Division 1 241 Y bus. Most, but not all, of the safety-related equipment in one train (Division 1) was unable to respond to the transient until bus 241Y was restored to operability, which took over 11 hours.

The complications included the following:

- The High Pressure Core Spray and the Reactor Core Isolation Cooling (RCIC) systems automatically tripped on high water level.
- RHR pump 2B failed to start due to the false undervoltage signal on the 241 Y bus.
- RCIC system suffered a water hammer event.
- RCIC outboard check valve position indication displayed an open indication when the valve was shut.
- Condenser hotwell reject valves did not adequately control level leading to a leak from the Condensate Storage Tank.
- Various operator performance deficiencies.

This event was modeled as an initiating event with loss of reactor feedwater with complications resulting from component failures and operator actions as described above. The mean and CCDP for this event is 1×10^{-5} . The confidence interval is a 5% uncertainty bound of 3×10^{-7} and 95% uncertainty bound of 4×10^{-5}.

SDP/ASP Comparison: No SDP analysis was performed since the event at LaSalle was an initiating event. The ASP risk analysis is consistent with the licensee's risk analysis.

The ASP analysis can be found at ML041210839. If you have any questions about the analysis, please contact Eli Goldfeiz (415-4439).

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REMARKS

Transmittal of Final ASP Analysis

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